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PHYSICAL CHARACTERISTICS OF RESEARCH POOL-TYPE REACTOR IRT-2000 WITH TEST-LOOP CHANNEL

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I. Introduction

Reactor IRT-2000 at the Heat and Mass Transfer Institute of the B.S.S.R. Academy of Sciences was put in operation in May, 1962. The distinctive feature of reactors with water as coolant and moderator of the IRT-type (the detailed description of such a reactor can be found in [1,2]) is the simplicity of the construction, high reliability and safety in operation, and at the same time such reactors give wide opportunities for conducting many experiments on radiation chemistry, nuclear physics, biology[1]).

Reactor IRT-2000 is a standard one. In its design no test loops are supposed. The experimental possibilities of the IRT-type reactor widen essentially when a test loop channel is installed in the reactor core. In connection with this, the study of physical characteristics of reactor IRT with a test loop channel has been made.

1/ A brief characteristic of works which are being run on the experimental reactor channels is given in Appendix I.

In this paper the calculation and experimental results on determination of the physical characteristics with the test loop channel (critical masses, neutron flux distribution, excess reactivity) are shown, and the changes in design of some reactor units due to the test loop channel installation are described.

2. Changes in design of some reactor IRT-200 units carried out for test loop installation

To provide the possibility of carring out works on the loop installations a hole is made 90 mm in diameter in the centre of the core, that permits the installation of loop channels of several types. Creating the duct in the centre of the core has led to changes in construction of some of the assemblies situated in the reactor pool.

In the lower lattice on the reactor-vessel axis a through opening 27 mm in diameter is made where the lower end of the loop channel can be fixed. In the upper lattice two mutually perpendicular crosspieces separating four central fuel assemblies are cut off (Fig.1).

The reconstructed upper lattice provides reliable spacing of the central flue assemblies and allows the installation of the test loop channels up to 85 mm in diameter in the reactor core. To extract the test loop channel from the core, some changes are made in the design of an assembly-holder and an umbrella deflector. Two central knives of the assembly-holder are shortened by 240 mm (Fig.2). The pressing facility fastened by two adjacent knives of the assembly-holder is fixed instead of the remote parts of the knives. The hole for the central experimental channel in the umbrella-deflector is enlarged up to 90 mm.

To provide installation and reloading of the test loop channel the reactor top platform was reconstructed. On reconstruction of the platform its steel framework was partially cut, and to provide sufficient rigidity and firmness the entire platform was therefore reinforced by two brackets fastened to the pool wall.

The technological test loop equipment (pumps, heat exchangers, compressors, etc), main loop communications and auxiliary systems (systems of purification, filling, etc) are situated in a special room built in the zone of the horizontal experimental channel 10. The

718

walls and the ceiling of the technological test loop room are made of heavy concrete with specific density 4.5 ton/m^3. The room has special ventilation and sewerage systems, cold water, hot water and compressed air supply from the corresponding reactor systems. For the loop-channel storage in the reactor pool the removable brackets are designed, on which the loop channel can be hunged up. The brackets are put in the grooves fastened to the reactor pool wall.

Piping of the main loop is located in the inclined channels in the concrete biological shielding of the reactor. The loop installation will be operated from the control panel situated above the technological room of the loop (Fig.3).

3. Calculations of the physical characteristics

As a design model of the loop channel was taken the worst (as to its effect on the reactor reactivity) homogeneous mixture of the steel and water.

It was assumed that there was 2.8 kg of stainless steel (IXI8H9T) in the cylindrical channel 75 mm in diameter, that approximately corresponds to the weight of steel in some loop channel constructions.

To put the loop channel into the core centre, it is necessary to take out four fuel assemblies (Figs.1,2). Vacant space round the channel can be filled either by some moderator displacers or by figured fuel assemblies.

Calculations were carried out in the 5-zone cylindrical geometry: 1st zone - loop channel, 2nd zone - zone of displacers, 3d zone - core, 4th zone side reflector in core cells, 5th - side reflector (reactor pool water).

Transition from the real core to the cylindrical one was based provided that the geometric parameter of the core was constant. The height of the design model was taken equal to that of the core.

The 3-group method similar to that described in [3] was chosen. Three-group neutron balance equations derived on the basis of diffusion approximation of the kinetic equation have the form:

$$D_j \nabla^2 \Phi_j(r) - \Sigma_j \Phi_j(r) + \sum_{k=1}^{j-1} \Phi_k(r)\Sigma_{jk} + \chi_j Q_f(r) = 0; \quad Q_f(r) = \sum_{j=1}^{3} (\nu_f \Sigma_f)_j \Phi_j(r); \qquad (1)$$

where Φ_j is the j-th group integral flux; χ_j is the fission spectrum integrated over the j-th group interval; Σ_j is the total cross-section for the j-th group; Σ_{jk} is the cross-section of slowing-down from the j-th group to the k-th one. The rest designations are gene-

778

rally accepted.

To separate the effects due to the influence of the fast neutron leakage and resonance absorption of the slowing-down neutrons on the critical mass of the reactor, the energy interval was divided into groups with the following boundaries: the first j=1, 300 kev$<E<\infty$; the second j=2, $E_b < E <$ 300 kev; the third j=3, $0<E<E_b$; (E_b is the conventional boundary between the Maxwell and Fermi spectra defined by [4]).

Average group cross-sections for the core and reflectors except uranium and water cross-sections are obtained by reducing the 10-group system of constants described in detail in [5] to the three-group one, using the following formulae: $\Sigma_{a1}=(\sum_{e=1}^{5}\Phi_e\Sigma_{ae})/\sum_{e=1}^{5}\Phi_e$;

$$\Sigma_{12}=(\sum_{e=1}^{5}\Phi_e\sum_{K=6}^{9}\Sigma_{eK})/\sum_{e=1}^{5}\Phi_e\ ;\quad D_1=(\sum_{e=1}^{5}D_e\Phi_e)/\sum_{e=1}^{5}\Phi_e\ ;\quad (\nu_f\Sigma_f)=(\sum_{e=1}^{5}(\nu_f\Sigma_f)_e\Phi_e)/\sum_{e=1}^{5}\Phi_e\ ; \quad (2)$$

where e is the number of a group in the tenth-group approximation.

Lower boundary energy of the fifth group in the 10-group system is 300 kev and that of the nineth group is E_b.

The similar formulae were used for defining the 2nd group average cross-sections. As a flux weight function the neutron spectrum of the reactor IRT obtained for the bare core by the multigroup method was used. The reflector effect on the flux spectrum was taken into account by changing real core sizes to equivalent ones. Such assumptions lead to the following form of the flux spectrum:

$$\Phi_e=(\chi_e+\sum_{K=1}^{e-1}\Phi_K\Sigma_{Ke})/(\Sigma_e+\varkappa^2D_e)$$

More strict approach was used to the uranium and water cross-sections averaging because the fast neutron leakage and resonance absorption on the nuclei U^{235} and U^{238} had the strongest effect on the reactivity of the water cooled-moderated reactor IRT. It was assumed that uncertainties in the description of the neutron-water interaction lead to the errors only in the value of the diffusion coefficient for the fast neutron group. So the values of Σ_{aj}, Σ_{12}, Σ_{23}, D_2 for the water were measured using formulae of form (2) and the diffusion coefficient for the fast neutron group D_1 was defined from the requirement of the group neutron symbolic age coincidence with the experimental value of the symbolic age for the fission neutrons in water. The experimental value of the symbolic neutron age at the energy of the indium resonance τ = 27.68 cm^2 [6] was used. After introducing the correction for the neutron age, equal to:

$$\Delta\tau=\int_{E_b}^{1.44} D(U)\,dU/_{3}\Sigma_s(U)=0{,}92\ sm^2;$$

we have

$$\tau+\Delta\tau=D_1\chi_1/(\Sigma_{12}+\Sigma_{a1})+D_2(\chi_2+\chi_1\Sigma_{12})/(\Sigma_{23}+\Sigma_{a2})\ ; \qquad (3)$$

718

and D_1 can be obtained from (3).

To compose the constants for the reactor pool water zone the neutron leakage through horizontal experimental channels was calculated by the method suggested in /7/ .

Group cross-sections for uranium isotopes were estimated taking into account the self-shielding effect in the narrow-resonance approximation /8/ .Average microscopic cross-sections for the thermal neutron group were taken from /4/ .When calculating the thermal macroscopic cross-sections the flux distribution over the fuel assembly was taken into account.The thermal neutron flux depression in the fuel element was denined by using the results of the method of successive generations /9/ and in the moderator and in the structural materials by using the diffusion approximation.The system of multigroup equations (2) was solved numerically by the electronic computer "Ural-1" by the method of source interactions.

The calculation results of absolute neutron fluxes with power of 1000 kwt and critical masses of reactor IRT with the loop channel surrounded by various displacers are shown in Fig.4 and in Table II.In these calculated data on critical masses and excess reactivity the thermal graphite column effect was taken into account.Estimations of the thermal column effect were made using a slab model and taking into account the lead screen."Weight" of the thermal column has a maximum in the case of full packing of the core cells by the fuel assemblies and is equal to 1.38% ΔK_{eff} .In this case of the core consisting of 26 fuel assemblies surrounded by the water reflector the minimum "weight" is equal to 0.3% ΔK_{eff} .The results of calculations are in rather good agreement with the experiment (See Table I).

Consideration of the possible errors in calculating the critical masses (uncertainties in measurement of the microscopic cross-sections,symbolic neutron age,etc) allowed the indication of the interval of maximum errors in the critical mass calculations as ± 10%.

4.Experimental study of reactor IRT-2000 physical characteristics[2]

2/The critical mass experiments were conducted under the guidance of Yu.G.Nikolayev (the Kurchatov Institute of Atomic Energy)

718

To prove the possibility of the test loop channel installation in a reactor of the IRT-type, the critical experiment on determination of critical loading of the reactor with the loop channel was conducted. The loop channel imitator made of stainless steel IXI8H9T had a weight of 2.63 kg and was located in a water cavity 90 mm in diameter. The water cavity was formed by the aluminium displacers which were put in the four central cells of the core (Fig.1). Criticality was reached when 38 fuel assemblies (Fig.5d) were loaded and when the whole automatic control rod was inserted into the core. The extrapolation of the reversed counting curves has shown that the criticality can be achieved when loading of 37.3 fuel assemblies (4.76 kg of U^{235}) when the automatic control rod was taken out.

The loop channel imitator effect on the reactor reactivity by the critical experiment without a steel imitator in the water cavity. The criticality was reached when the core loading was 37 fuel assemblies (Fig.5c) and the automatic control rod was partially inserted into the core. The extrapolation of the reversed counting curves has shown that compensation could be reached when loading of 36.3 fuel assemblies (4.63 kg of U-235). One can see from the experimental results that the steel loop-channel imitator, inserted in the water cavity in the centre of the core, results in increasing the fuel loading only by one fuel assembly, that is by 128 g of U-235.

Besides the determination of the critical loadings, the aim of the experiment conducted without the loop channel in the water cavity was to study the possibility of providing an effective neutron "trap" in the reactor IRT core.

The relative thermal neutron flux "peak" in the water cavity was measured by activations of copper wires which were put into the central experimental channel and among the fuel elements in the adjacent fuel assemblies. The measurements have shown that the thermal neutron flux in the centre of the water cavity surrounded by the aluminium displacers was higher than in the nearest fuel assemblies by the factor of 2.5.

The neutron "trap" effect on the reactor reactivity was defined by the results of the critical experiment with the cylindrical aluminium displacer in the core, placed instead of the central water cavity. In this case the critical loading consisted of 34 fuel assemb-

718

lies (Fig.5b) or 4.34 kg of U-235.Comparison of the results of two critical experiments with the central water cavity and with that occupied by the cylindrical aluminium displacer has shown that creation of the water neutron "trap" in reactor IRT increases the critical reactor mass,at least,by 2.3 fuel assemblies.

To define the reactor composition effect on the reactor excess reactivity for a mixed water-graphite reflector many experiments were conducted.Without the loop channel,the reactor criticality was reached by the loading of 26 fuel and 2 graphite assemblies installed on the thermal column side (Fig.5a) and by the automatic control rod partially inserted into the core ("weight" of the inserted part of the rod was equal to 0.14%).The subsequent loading into the core was made by two graphite assemblies at each step.The experimental results are shown in Fig.5.From the comparison of the excess reactivity one can see that the share of each subsequent assembly increases.The maximum "weight" of one graphite assembly does not exceed 0.265%.When the whole water reflector was replaced by the graphite one,the excess reactivity increases by 3.5%.

The results of the critical experiments are used to verify the calculation method.

D i s c u s s i o n

The results on critical masses (Table I,Fig.5),excess reactivity (Table II,Fig.5) and neutron fluxes (Fig.4) can be used to estimate the experimental possibilities of reactor IRT-2000 with a loop channel.

The results of calculations and critical experiments show the principal possibility of the loop channel installation in the core centre.The weight (2.6 kg of steel) of constructional materials of the loop channel is equal to only one fuel assembly (Fig.5d),that provides the possibility of constructing ~everal loop channels.

The material of displacers surrounding the loop channel has a considerable effect on the critical mass.Among the variants considered (water,graphite,berillium,berillium-oxide,aluminium) the smallest critical mass (34.0 fuel assemblies) was obtained when the loop channel was surrounded by berillium.The largest one (39.7 fuel assemblies) was obtained when the loop channel was surrounded by water.Thus,the loop channel located in the core increases the critical reactor mass according to the material surrounding the channel

718

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by 8 ÷ 14 fuel assemblies (with a water core reflector). A specific choice of the material around the channel depends on the aim of the loop tests. So in many experiments (e.g. radiation chemistry experiments) where the maximum thermal neutron flux is necessary, the loop channel cavity must be surrounded by water (Fig.4). It should be noted that in this case the excess reactivity is not enough even for compensation of poisoning[3] (See Table II) and it is therefore necessary to replace some fuel assemblies by those containing more U-235 in the fuel elements. If the main demand is a maximum campaign, the loop channel should be surrounded by either fuel assemblies or graphite or berillium displacers. As one can see from Table II, the loop channel reactor taking into account poisoning has relatively small resource. However, it should be taken into account that after the completion of loop experiments the loop channel can be taken out from the core and its zone can be again filled by the fuel assemblies.

The filling of the cavity in the core centre by water allows the creation of an effective neutron "trap" in reactor IRT. Surrounding the water cavity by fuel assemblies allows the thermal neutron flux of about 1.10^{14} neutron/cm^2 sec to be obtained in the neutron "trap" when operating with power of 2000 kwt. The creation of the "trap" increases the critical mass approximately by 5 fuel assemblies.

Conclusions

1. The results of the critical experiments and neutron physical calculations have shown that it is possible to install a loop channel containing about 3 kg of steel in reactor IRT.

2. The results of the work allow the choice of the optimum material around the loop channel to reach a maximum compaign of the reactor with the loop channel or a maximum thermal neutron flux.

3. The filling of the cavity in the core centre by water allows to have a neutron "trap" with a greatly high thermal neutron flux: $1.6.10^{14}$ neutron/cm^{2}sec with power of 2000 kwt in reactor IRT.

4. The changes in the reactor construction due to the loop installation can be rather easily performed and recommended for new reactors of the IRT-type to be built.

3/As was shown experimentally, a deacrease in the excess reactivity of reactor IRT operating at 1000 kwt due to poisoning is 1.95%

718

R E F E R E N C E S

1. Goncharov V.V., Nikolayev Yu.G., Babulevich E.K. and Chernilin Yu.K., Experimental pool-type 2000 kwt reactor (Eksperimental'nyi reaktor basseinogo tipa moshchnost'yu 2000 kwt). 2nd United Nations Int. Conf. on the Peaceful Uses of Atomic Energy, Geneva, 1958 (Russian ed. v.2, Atomizdat, 1959).

2. Bulkin Yu.M. and Volkov E.N., A nuclear reactor for research aims (Yadernyi reaktor dlya eksperimental'no-issledovatel'skikh tse - lei). Journ of Physical Engng. No. 10, 1958.

3. Deutsch R.W., Nuclionics 15, (1957), 47.

4. Gordeyev and Kardashev D.A., Nuclear and physical constants (yaderno-fizicheskiye konstanty). Izd. AE SC USSR (1960), 261.

5. Naumov V.A., The 10-group method for calculating the neutron flux attenuation in the thermal and biological shielding (IO gruppovoi metod dlya raschoyta oslableniya potoka neitronov v teplovoi i biologicheskoi zashchite). Thesis (1963).

6. Munu A. and Pontecorvo B., Spatial distribution of neutrons in hydrogeneous media containing bismuth, lead and iron. Can. I. Res. A; 254 (1947), 157-167.

7. D.I. Behrens, The effect of holes in a reacting material on the passage of neutrons. Proc. of Phys. Soc. (1949), 607.

8. Shikhov S.B. and Abagyan A.L., Collection of works " Theory and Methods for calculating nuclear reactors" (teoriya i metody raschoyta yadernykh reaktorov). Gosatomizdat (1962), 200.

9. Stuart G.W. and Woodruff R.W., Nucl. Sci. and Engng. 3(3), 1958, 339-373.

718

CRITICAL MASSES OF REACTOR IRT-2000

TABLE I

N	Critical mass number of fuel assemblies/ kg of U-235		Reactor type		Note
1.	Calculation	Experiment	in core cells (Nos. of cells)	outside core vessel	
1.	20.3 / 2.57	less than 20[4]/ 2.47	graphite-28	water	Core without loop channel and displacers
2.	28.0 / 3.55	25.8 / 3.27	graphite-2 water-20	Water with horizontal experimental channels	- " -
3.	38.0 / 4.85	37.3 / 4.76	water-11	- " -	Core with loop channel and aluminium displacers

4/ From the data of 1 .

PREDICTED CRITICAL MASSES AND EXCESS REACTIVITIES OF LOOP CHANNEL REACTOR IRT

TABLE II

N.	Filling of displacer zone	Reflector in core cells	Critical mass of fuel assembly/kg of U-235	Excess reactivity in total loading of core, %
1.	fuel assemblies	graphite	30.9/3.94	5.8
2.	fuel assemblies	water	34.2/4.36	5.8
3.	berillium	water	34.0/4.34	5.1
4.	berillium oxide	water	34.4/4.39	4.8
5.	graphite	water	35.3/4.50	4.2
6.	aluminium	water	38.0/4.85	2.1
7.	water	water	39.7/5.07	1.4

718

Appendix I

A BRIEF CHARACTERISTIC OF INVESTIGATIONS ON EXPERIMENTAL REACTOR CHANNELS

At present all the horizontal reactor channels are mastered and the following experiments are being carried out on them[5].

Channel No.1 γ - coincidence measurements in the (n, γ) reaction by scintillation spectrometers.

Channel No.2 Investigation of space distribution of a reactor neutron flux in various organic media.

Channel No.3 Neutronographical study of magnetic structures and formfactors of intermetallic manganese compounds by polarized neutrons.

Channel No.4 γ - ray spectra measurements in a (n, γ) reaction on pure isotopes.

Channel No.5 Study of atomic oscillation frequency spectrum in solid crystal lattice by neutron scattering.

Channel No.6 Study of coincidence between γ -rays and conversion electrons from (n, γ)-reaction and that of conversion electron spectra.

Channel No.7 Study of kinetic characteristics of fission fragments with fission induced by the polarized neutrons.

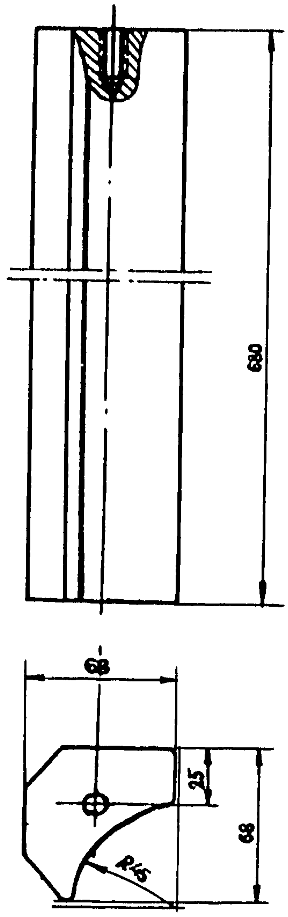
Channel No.8 Neutrongraphical study of the inversion degree of ferrite systems of various composition.

Channel No.9 Study of the effect of intermediate neutrons in small doses on physiological functions and metabolism of animals and that of relative biological effectiveness of intermediate neutrons on microorganisms.

Horizontal reactor channels are studied by the Institutions and Departments of the B.S.S.R. Academy of Sciences and Byelorussian State University.

5/ The channel No.10 which has the smallest neutron flux at the exit is an exeption.This channel ends at the loop installation room and in the nearest future will not be of use.

718



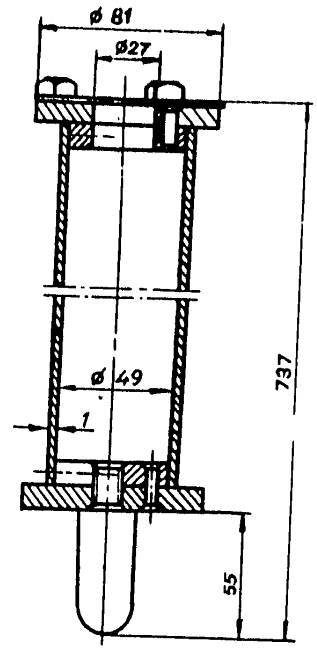
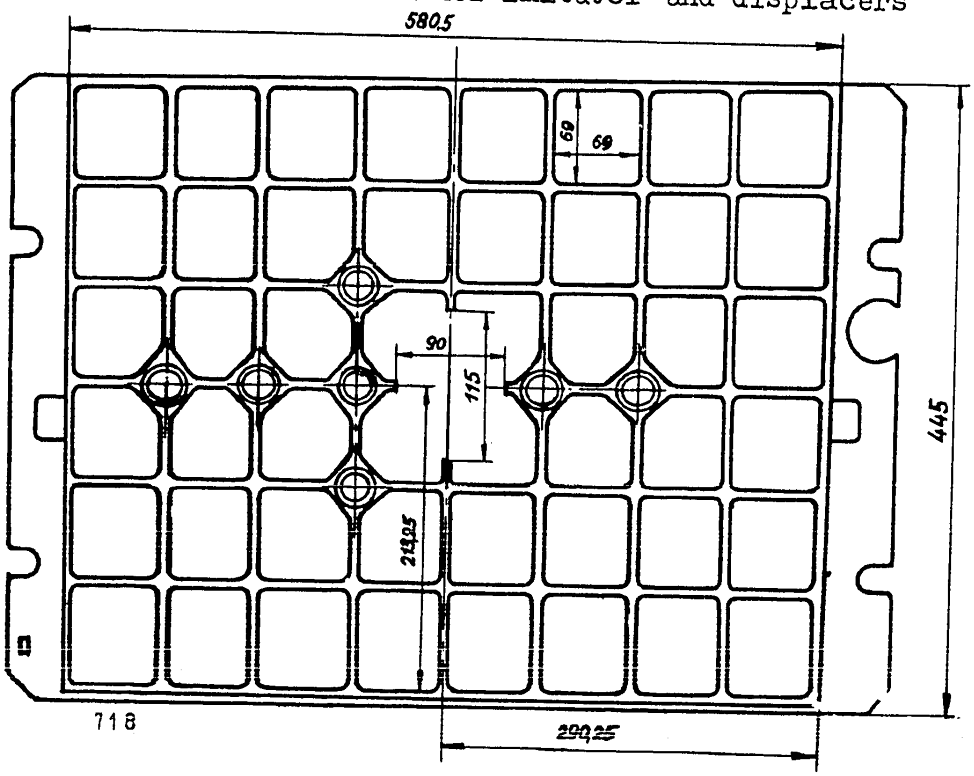


Fig.1

Reconstructed upper lattice,
construction of loop
channel imitator and displacers



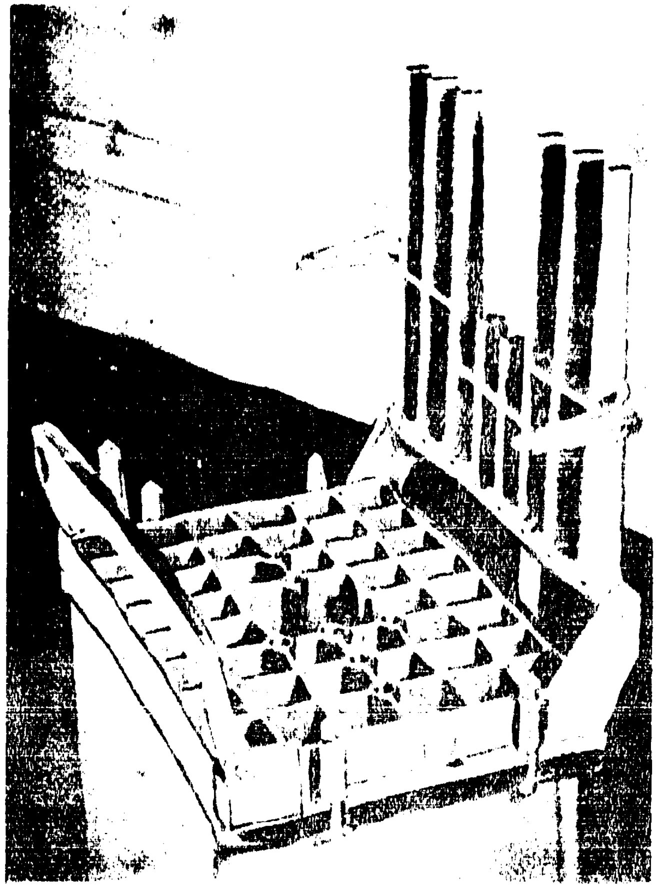


Fig.2 Reconstructed reactor core

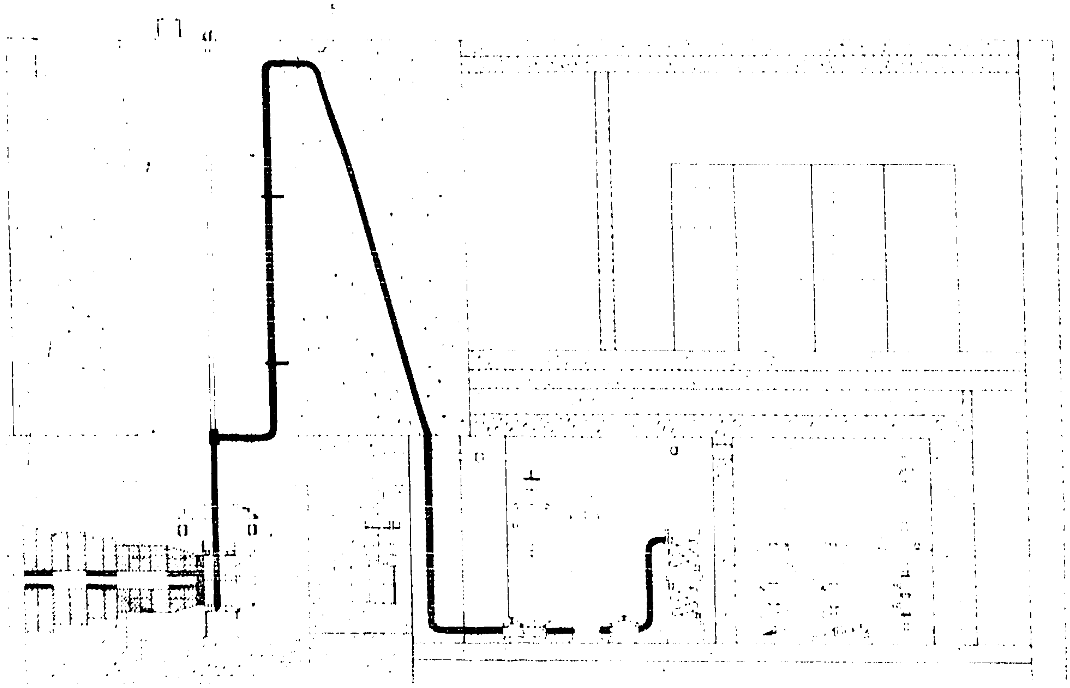


Fig.3 Disposition of loop installation at reactor IRT-2000

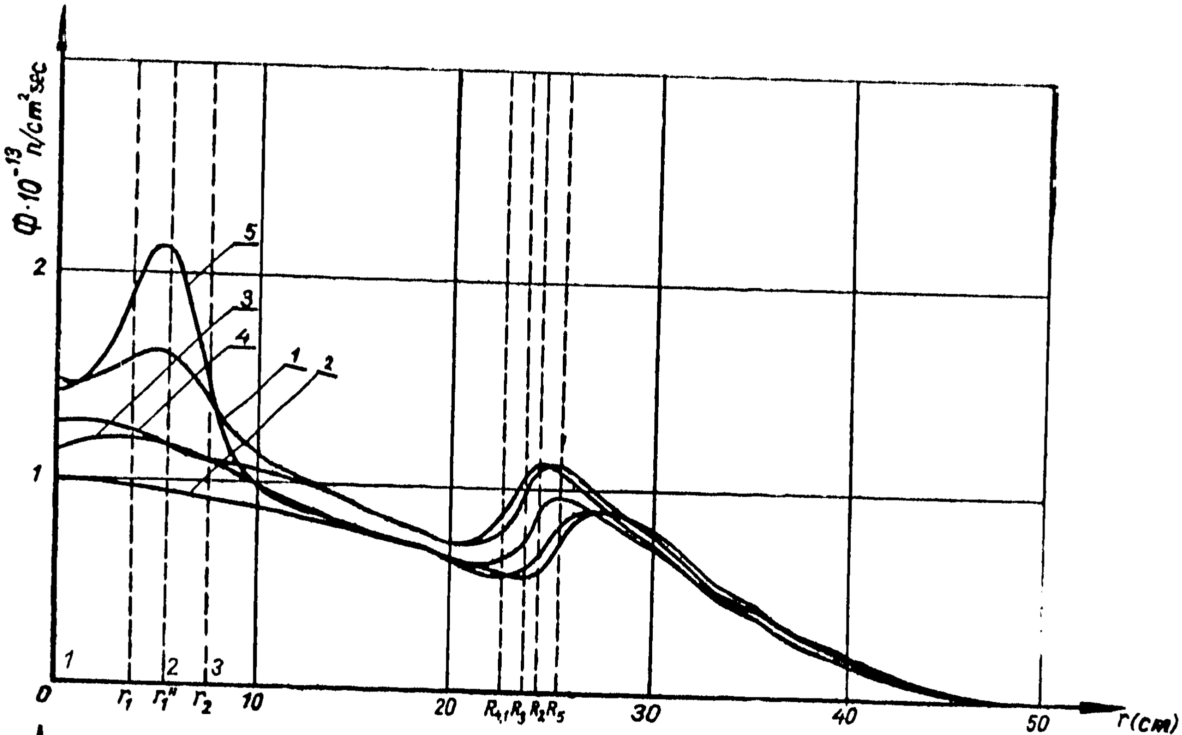
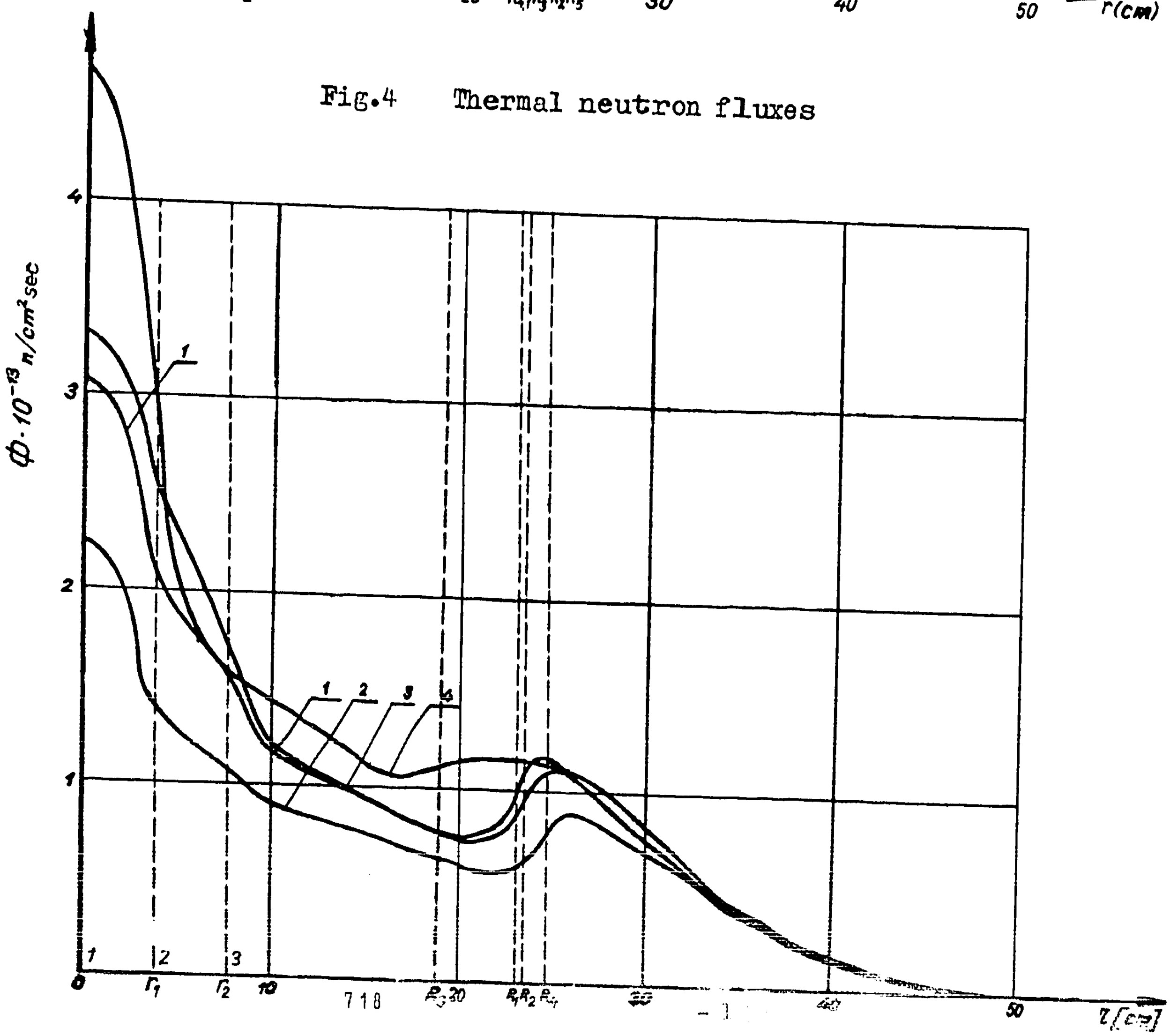


Fig.4 Thermal neutron fluxes



718

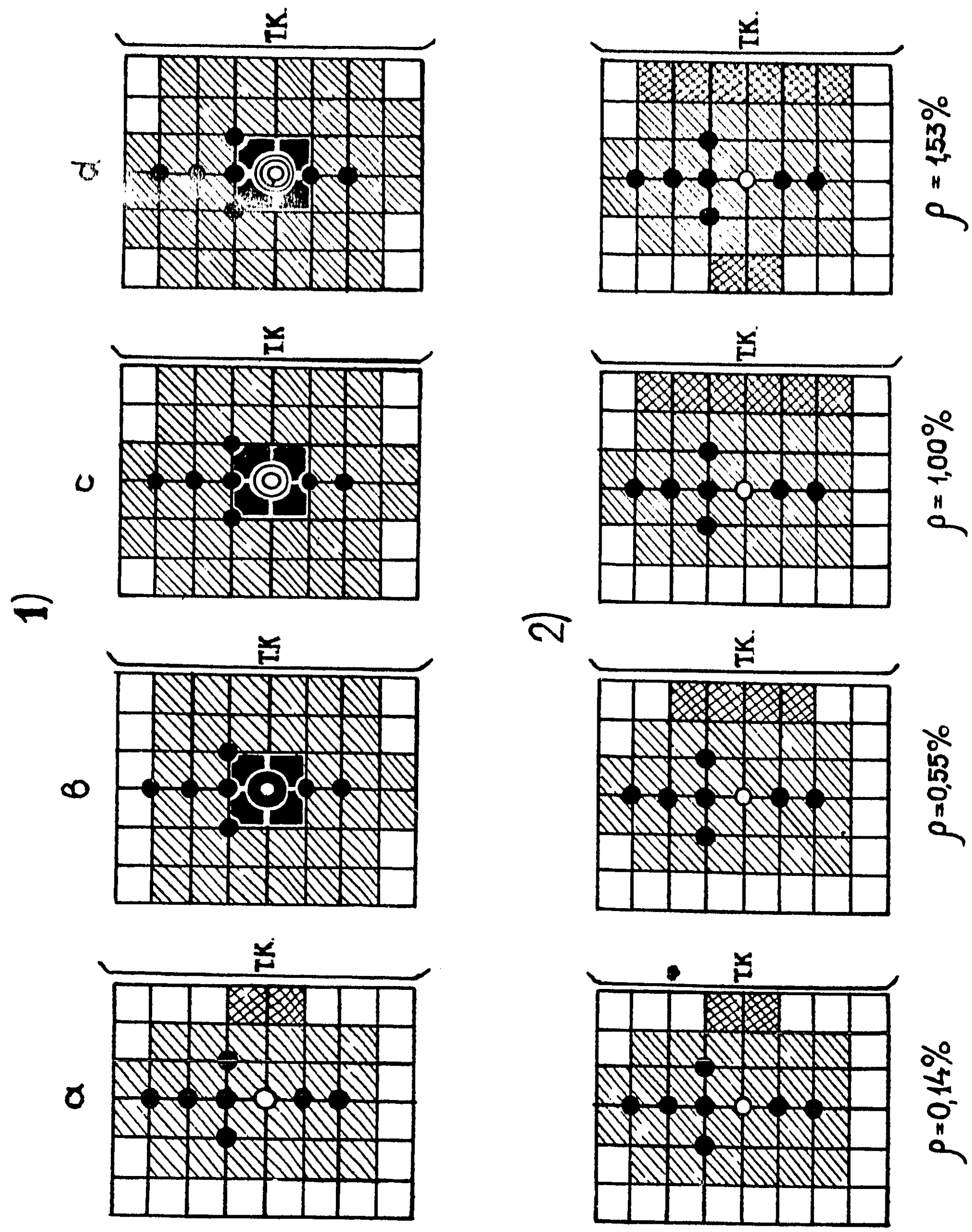


Fig.5 Core loading diagrams

718